

NON-PUBLIC?: N

ACCESSION #: 8809290344
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Catawba Nuclear Station, Unit PAGE: 1 OF 7

DOCKET NUMBER: 05000414

TITLE: Reactor Trip Due To Operator Aid Computer Training and Graphics Design
Deficiency

EVENT DATE: 09/15/88 LER #: 88-019-01 REPORT DATE: 09/15/88

OPERATING MODE: POWER LEVEL:

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

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COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE TO NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On May 27, 1988, at 1403:21 hours, an undervoltage condition occurred on the 240/120 VAC Auxiliary Control Power System distribution panel 2KXPB. This undervoltage condition occurred while cycling the alternate source to KXPB breaker in an attempt to clear an indicated alternate source undervoltage. The indicated undervoltage condition was noticed while isolating the 2KXPB inverter for corrective maintenance. The panel's power supply had previously been swapped to the Alternate Source to KXPB supply from regulated AC power source, 2RDB. Upon noticing the Low Alternate Source Voltage indication on the manual bypass switch, 2KXMB, the involved Nuclear Operations Specialist (NOS) requested assistance from Control Room personnel. A Control Room Operator (CRO) utilized the Operator Aid Computer (OAC) graphics to verify that a low voltage was indicated. The CRO recommended that the NOS cycle the Alternate Source to KXPB Breaker to clear the undervoltage condition. Cycling the breaker caused an approximate 5 second loss of power to the loads supplied by the KXPB distribution panel, one of which was the control power to Main Feedwater Pump Turbine (CFPT) 2B. This resulted in a decrease in CFPT 2B speed, low Steam

Generator levels, and a subsequent automatic Reactor trip. The Unit was at 100% power at the time of this incident.

This incident has been attributed to a Management Deficiency, due to a lack of training given to Reactor Operators on the Operator Aid Computer (OAC). Also assigned as a contributing cause is a Design Deficiency, due to insufficient human factors consideration because of misleading information which was contained in the OAC graphics software. The health and safety of the public were unaffected by this event.

END OF ABSTRACT

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BACKGROUND:

Inverter EIIS:INVT! 2KXIB EIIS:EC! is a 240/120 VAC power supply for important, but non-safety related, plant loads. The inverter converts 125 volt D.C. power from distribution panel EIIS:PL! 2CDB to 240/120 VAC power and delivers it through autostatic transfer switch EIIS:CS! 2KXAB, and manual bypass switch EIIS:CS!, 2KXMB, to distribution panel 2KXPB, from which those plant loads are supplied.

The Inverter may be taken out of service and the plant loads still supplied by aligning an alternate AC power supply from regulated power supply bus 2RDB to 2KXPB through the manual bypass switch, 2KXMB. This switch assures continuous power is supplied to 2KXPB by a "make before break" configuration. Because of this type transfer, the power supplies must be synchronous, and have the appropriate voltage and frequency. Verification of these parameters is available through the use of metering and alarm lights on the inverter and manual bypass panels.

In the event of an undervoltage from the inverter, autostatic transfer switch 2KXAB will automatically supply power from the alternate source of power 2RDB, to the 2KXPB distribution panel.

DESCRIPTION OF INCIDENT:

On April 25, 1988, Work Request 40247 OPS was written to request IAE to repair a fan in the Auxiliary Control Power System Inverter 2KXIB. On May 27, 1988, a Nuclear Operating Specialist (NOS) was assigned the task of removing 2KXIB from service for those repairs. He reviewed the task and determined that he would place the inverter in total bypass by switching manual bypass switch 2KXMB from normal operation to the "Alternate Source to Load" position per Operations procedure 0P/2/B/6350/09, 125 VDC - 240/120 VAC Auxiliary Control Power System, Enclosure 4.17. He then would deenergize the inverter using the same procedure.

At approximately 1350 hours, the NOS completed the first steps of the isolation, and the 2KXPB plant loads were being supplied from 2RDB. At this point, the NOS noticed that AC voltage was approximately 117 volts and frequency was correct, but an alarm light, Alternate Source Low Voltage, was activated. He compared it to the same alarm light on Manual Bypass Switch, 2KXMA, and the corresponding Unit 1 switches. The 2KXMA undervoltage alarm was also illuminated but the Unit 1 alarms were not.

Because of the undervoltage indication, the NOS stopped all work on the procedure and called the Control Room for assistance. A Control Room Operator (CRO) used the Operator Aid Computer (OAC) EIIS:CPO! graphics option to verify the undervoltage. The OAC showed that the lines from 2RDB to 2KXMB and 2KXAB were deenergized and that the breaker from 2RDB to 2KXAB was open. The NOS and CRO verified the breaker labeling as "Alternate Source To 2KXIB", compared it to the OAC schematic placement of the breaker and determined that cycling the breaker would not cause any adverse effects (see Enclosure 6.1). The CRO advised the NOS to cycle the breaker in an attempt to clear the undervoltage condition.

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At 1403:16 hours, the NOS opened breaker 2RDB-FOIC, labeled Alternate Source to 2KXIB, which deenergized the loads on 2KXPB. Several plant indications were deenergized and power was interrupted to Main Feedwater Pump Turbine (CFPT) 2B controls. The CFPT 2B speed decreased from 5105 rpm. This created a decrease in Main Feedwater flow and pressure and also caused Steam Generator (S/G) levels to decrease. As flow through the pump reduced to the low flow setpoint, CF Pump 2B Recirculation valve, 2CF13, opened automatically.

The CRO saw the transient occur on CFPT 2B and took manual control of CFPT 2B in an effort to regain control of the pump and S/G level. Automatic controls raised the speed of CFPT 2A to its maximum setpoint in an effort to compensate for the reduction in feedwater flow.

At 1403:18 hours, the NOS reclosed 2RDB-FOIC and power was restored to those components supplied by 2KXPB. CFPT 2B stabilized its speed at approximately 3350 rpm, but S/G levels continued to decrease.

At 1404:32:821 hours, S/G 2D level channel 4 decreased to the low low Reactor trip setpoint. At 1404:32:827 hours, S/G 2D Channel 3 reached the low low Reactor trip setpoint and the Reactor tripped on 2 of 4 low low S/G channels in one S/G. The Main Turbine Generator tripped due to a Reactor trip. Both Motor Driven Auxiliary Feedwater (CA) Pumps started. As levels in the S/G 2B decreased to the low low setpoint, the Turbine Driven CA Pump started on 2 of 4 channels

low low level in two S/Gs. CF Pump 2A went into recirc at this time. Over the next several seconds, all other S/G levels decreased below the trip setpoint as well.

At 1408 hours, the CROs reset the CA autostart signal and secured the Turbine Driven CA Pump to regain control of the CA Flow Control valves and to minimize cooldown. At 1420 hours, CFPT 2B was manually tripped to minimize cooldown. A Hotwell and Booster pump were also manually tripped. A CRO noticed at 1450 hours, that S/G 2B Upper Sample Inside Containment Isolation valve, 2NM197B, was not fully closed. They then verified that 2NM210A was closed to isolate the penetration, and later tagged the valve in that position to comply with Tech Specs. By 1510 hours, S/G level alarms had cleared and the Unit was stabilized in Mode 3, Hot Standby.

CONCLUSION:

This incident is classified as a Management Deficiency, due to the lack of training given on the OAC to the involved Operators. The decision to cycle the breaker from the alternate source to the manual bypass switch came only after attempting to evaluate the situation with the available information from the OAC. Correct interpretation of this particular OAC graphic could only be made through a detailed study of the logic and components represented on the graphic display. The opportunity to become fully acquainted with the content of this program has never been presented to the Operators.

Any training given on the OAC should highlight limitations in its usefulness. In this case, 117 VAC was present during the undervoltage which was indicated on the

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graphic display. The 117 VAC decreasing setpoint for the undervoltage alarm changed the color of the graphic display segment to indicate de-energized. Training should point out that "undervoltage" does not necessarily mean "zero voltage" is present. Additionally, it should be emphasized that the graphic software often deduces output displays from logic statements based on other inputs. For example, a breaker position and voltage indication in one area of the graphics will affect indicated voltage downstream of the breaker on the graphic. An error in one segment may be passed logically to several other segments.

This incident has also been assigned as a Design Deficiency, because of the misleading information which was contained in the OAC graphics software. Had this information been clear, the first attempts to investigate the situation would likely have been successful and the breaker not cycled. However, the CRO and several other plant personnel believed that the OAC portrayed the 2RDB to

2KXPB breaker on the graphic. This "breaker", in actuality, was a part of the manual bypass switching circuit.

A review of previous incidents revealed a similar incident involving the unnecessary cycling of a control power breaker causing a Reactor trip (see LER 414/87-07). The corrective actions taken for that incident, including discussion of the incident with the personnel involved, the issuance of a shift update and the discussion of the event with the Shift Supervisor, did not prevent this event, because the event was not attributed to a training error. Another previous incident in which a reactor trip due to a training deficiency occurred, involved improper training on the use of the reactor trip breakers and could not have prevented this incident (see LER 414/86-22). A third incident involved training on Turbine Generator rate of loading controls (see LER 413/87-06).

There have been two previous Reactor trips attributed to design deficiencies, although neither involved the Operator Aid Computer graphics information.

During this incident, several components did not exhibit adequate performance. 2NM197B did not indicate closed following the CA autostart. Work Request 40540 OPS was written to investigate and repair this valve. The penetration will remain isolated by keeping the outside Containment Isolation valve closed until repairs are completed.

During the plant response to the trip, several alarms related to the Auxiliary Feedwater System were received. The Performance, system expert for the system will evaluate these indications and -determine the corrective actions.

Three Condenser Dump valves did not perform correctly during the incident. Valves 2SB21, 2SB24, and 2SB27 will be investigated under Work Requests 7491 IAE, 7492 IAE, and 7493 IAE, respectively.

Work Requests 40556 OPS and 40557 OPS have been initiated to investigate the low voltage condition indicated on 2KXMA and 2KXMB, which contributed to the incident.

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Labels on 2KXKA and 2KXMB alternate source to load breakers and the same labels for 2RDA and 2RDB will be corrected to indicate that the breakers supply power to 2KXPA and 2KXMB, as the corresponding Unit 1 Breakers.

SUBSEQUENT

(1) Control Room personnel stabilized the Unit in Mode 3, Hot Standby, following the Reactor trip.

(2) 2NM197B was isolated by closing and tagging 2NM210A.

(3) Operations personnel discussed the incident with the personnel involved.

(4) Operations personnel discussed the incident at the Shift Supervisors meeting on June 10, 1988.

(5) Operations personnel have issued a shift update to reemphasize the correct operation of the inverter manual bypass switches.

PLANNED

(1) Planning will investigate problems with 2KXHA and 2KXMB voltage conditions on Work Request 40556 OPS and 40557 OPS.

(2) Planning will investigate the actuators for 2SB21, 2SB24, and 2SB27 per Work Requests 7491 IAE, 7492 IAE, and 7493 IAE.

(3) Operations will generate an SPR to revise labels on Unit 1 and Unit 2 EPF Alternate Source breakers to correctly identify their function.

(4) Integrated Scheduling will insure that 2NM197B valve actuator is repaired under Work Request 40540 OPS.

(5) Production Support/Operator Training will evaluate the feasibility of implementing OAC training relative to its impact on presently allocated personnel resources.

(6) Performance will further evaluate the problems identified in the Transient Cycle Mini- Trip Report in the Auxiliary Feedwater System,

and initiate corrective actions as required.

(7) Operations will complete a review of the OAC graphics software for accuracy/ability to be easily and clearly understood. Concerns identified in this review will be forwarded to Design Engineering for resolution.

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SAFETY ANALYSIS:

Following the reduction in feedwater flow, automatic reactor trip was initiated

upon S/G D low-low level. CF isolation was automatically initiated upon reactor trip with low Tave (564 deg. F). Both motor driven CA pumps autostarted upon S/G D low-low level, and a turbine driven CA pump autostart signal occurred upon low-low level in two-out-of-four SIG's about 1 second after motor driven pump autostart. The redundant steam supply valves for the turbine driven CA pump, SA2 and SA5, opened within 8 seconds of the SSPS autostart signal. The reactor trip breakers opened within 72 milliseconds of the reactor trip signal and all of the control rods fell to the bottom of the core, reducing reactor power to decay heat level. The operators initiated manual reactor trip within 5 seconds of the automatic trip.

Reactor Coolant system temperature increased approximately 2 deg. F (to a maximum of 592 deg. F) prior to the reactor trip due to the decrease in feed flow, and stabilized at 540OF within 30 minutes post-trip, 17 deg. F from the no-load target of 557 deg. F. Pressurizer pressure decreased to a minimum of 1982 psig immediately post-trip, and stabilized at 2240 psig 30 minutes post-trip, 5 psi from the no-load target of 2235 psig. Pressurizer level reached a minimum value of 18% post-trip, and recovered to 24% within 30 minutes post-trip, 1% from the no-load target of 25%. Steam pressure increased to a maximum of 1125 psig immediately post-trip, and stabilized at 935 psig within 30 minutes post-trip, 155 psi from the no-load target of 1090 psig. S/G wide range level decreased to a minimum indicated value of 34% immediately post-trip (steam pressure correction of this value yields an actual minimum level of 43%), and within 30 minutes post -trip: S/G A recovered to an actual wide range level of 66%, S/G B recovered to an actual wide range level of 78%, S/G C recovered to an actual wide range level of 81%, and S/G D recovered to an actual wide range level of 68%.

The operators swapped from the 75 gpm to the 45 gpm letdown orifice, thereby avoiding a low pressurizer level and consequent automatic letdown isolation. With the exception of three valves which malfunctioned, all of the steam dump to condenser valves opened to dump steam to the condenser. Steam pressure increased from 1005 psig to 1025 psig immediately prior to the trip due to the reduction in feedwater flow. To mitigate the pressure increase upon reactor trip, S/G B PORV (SVI3), and S/G C PORV (SV7), opened and closed within 1 minute and 33 seconds and 42 seconds, respectively.

A Safety Parameter Display System (SPDS) heat sink alarm was generated as all four S/G narrow range levels dropped-off-scale immediately post-trip (heat sink alarm generated by all four S/G narrow range levels off-scale with auxiliary feedwater flow < 450 gpm), but cleared upon auxiliary feedwater flow delivery to the S/G's. The auxiliary feedwater flow rate was acceptable and well above the 450 gpm minimum cumulative flow to the SIG's as required by the Reactor Trip or Safety Injection Emergency Procedure, EP/2/A/5000/01. The reactor coolant was 58OF subcooled at the point of minimum reactor coolant system pressure. Adequate heat sink was available and maintained at all times for core decay heat removal.

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This event is bounded by the "Loss of Normal Feedwater Flow" transient as discussed in section 15.2.7 of the Catawba FSAR. The cooldown limits of 100 deg F per hour for the reactor coolant system and 200 deg. F per hour for the pressurizer were not exceeded. Integrity of the fuel cladding, reactor coolant system, and containment structure was maintained at all times. No radioactivity was released during the S/G PORV cycling and turbine driven auxiliary feedwater pump operation.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(vi).

The health and safety of the public was not affected by this event.

ATTACHMENT # 1 TO ANO #8809290344 PAGE 1 OF 1

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DUKEPOWER

September 15, 1988

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-19, Revision 1

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Revision 1 to Licensee Event Report 414/88-19 concerning a reactor trip due to operator and computer training and graphics design deficiency. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

/s/ Hal B. Tucker

Hal B. Tucker

LERPGL36.DI/lcs

Attachment

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ACCESSION #: 8809290352
